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J. F. Latkowski, S. Reyes, R. Meier

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J. F. LATKOWSKI, S. REYES, and W. R. MEIER
Lawrence Livermore National Laboratory, Livermore, California, USA
latkowski1@llnl.gov

Abstract

Lawrence Livermore National Laboratory (LLNL) is making significant progress in several areas related to the safety and environmental (S&E) aspects of inertial fusion energy (IFE). A detailed accident analysis has been completed for the HYLIFE-II power plant design. Additional accident analyses are underway for both the HYLIFE-II and Sombrero designs. Other S&E work at LLNL has addressed the issue of the driver-chamber interface and its importance for both heavy-ion and laser-driven IFE. Radiation doses and fluences have been calculated for final focusing mirrors and magnets and shielding optimization is underway to extend the anticipated lifetimes for key components. Target designers/fabrication specialists have been provided with ranking information related to the S&E characteristics of candidate target materials (e.g., ability to recycle, accident consequences, and waste management). Ongoing work in this area will help guide research directions and the selection of target materials. Published and continuing work on fast ignition has demonstrated some of the potentially attractive S&E features of such designs. In addition to reducing total driver energies, fast ignition may ease target fabrication requirements, reduce radiation damage rates, and enable the practical use of advanced (e.g., tritium-lean) fuels with significantly reduced neutron production rates, the possibility of self-breeding targets, and dramatically increased flexibility in blanket design. Domestic and international collaborations are key to success in the above areas. A brief summary of each area is given and plans for future work are outlined.

1. Introduction

Due to budgetary constraints, our efforts on the S&E aspects of IFE have been focused primarily on two designs: HYLIFE-II and Sombrero [1,2]. To some degree, these designs represent the extremes in IFE power plant design. Sombrero uses direct-drive, laser-driven targets. HYLIFE-II uses indirect-drive targets, which are driven by heavy-ion beams. While the HYLIFE-II target chamber is protected by a flowing, thick-liquid, the Sombrero uses a dry wall that is protected from the x-rays and debris by xenon gas. While these two designs cannot possibly encompass all possible features, they represent a large portion of the available parameter space.

In the following sections, we present an overview of our recent results, work in progress, and upcoming work. The main areas of work include neutron and photon transport, neutron activation, and safety and environmental assessments. We cover not only the Sombrero and HYLIFE-II designs, but we focus on general areas such as the driver-chamber interface, target materials selection, and the possibilities offered by fast ignition.

2. Computer Code System

The IFE Technology group at LLNL uses an extensive computer code system. These codes enable us to start with a conceptual design for an IFE power plant, perform neutronics (including radiation damage) assessments, calculate neutron activation and activity-related indices such as radioactive afterheat and waste disposal ratings, calculate time-temperature histories in the various power plant components, model accident scenarios, estimate radioactive releases, and calculate doses to the maximally exposed individual (MEI) for each accident scenario.

The TART Monte Carlo neutron and photon transport code is a workhorse of the IFE Technology Group [3]. While TART is quite similar to the more widely known MCNP code, TART is significantly faster due to its use of energy groups rather than continuous energy. With TART support available at the LLNL site, code improvements and bug fixes can be rapidly obtained. The TART package includes error-checking and geometry visualization tools, which are invaluable. Also included are tools for examination of cross sections and combination of results from multiple calculations (allows for easy use of multi-processing). The TART package may be obtained through the Radiation Shielding Information Computer Code Center (see http://www-rsicc.ornl.gov/SOFTWARE.html).

The ACAB code is used for calculation of neutron activation and activity-related indices [4]. ACAB uses energy-dependent neutron fluences, in an arbitrary group structure, in conjunction with user-specified material composition and irradiation/cooling schedules to calculate neutron activation. ACAB can calculate arbitrary irradiation/cooling schedules that include steady-state and pulsed irradiation. Activity-related indices calculated by ACAB include activity, biological hazard potentials, radioactive afterheat, contact dose rates, waste disposal ratings, and photon release rates. Additional indices can be easily added. ACAB currently uses the FENDL-A/2.0 cross section library, but can use any library that has been processed into the EAF format. ACAB is able to use arbitrary group structures; the only requirement is that the fluxes and cross sections use the same group structure. If the cross sections and the fluxes do not match, then the COLLAPSE processor may be used. For additional information about the ACAB code, send a request to the code author at jsanz@denim.upm.es.

The IFE Technology group at LLNL has adopted fusion-related safety computer codes that have been developed and/or improved by our Fusion Safety Program colleagues at the Idaho National Engineering and Environmental Laboratory (INEEL). These codes include a fusion-modified version of the MELCOR code, which is used for thermal hydraulics and aerosol transport [5]. A new module introduced by INEEL allows simulation of HTO transport and condensation. The CHEMCON code was developed by INEEL for long-term thermal analysis [6]. CHEMCON calculates time- and spatially-dependent temperatures in one-dimensional geometry. The code is able to consider radioactive afterheat and chemical reactions between tungsten, beryllium, and graphite with air or steam. LLNL modifications enable the code to better track the oxidation of graphite structures.

3. Safety Assessments

In safety assessments, our strategy has been to first identify severe accident scenarios and to make conservative assumptions for the radionuclide inventories and chemical forms of the major radioactive source terms. In the longer term, we plan to identify and analyze many different accident scenarios for a range of release and weather conditions.

3.1. HYLIFE-II

A conservative accident scenario was analyzed for the HYLIFE-II power plant design. In this scenario, we consider a complete loss-of-coolant accident (LOCA) along with the simultaneous failure of all beam tubes and 1 m^2 breaks in both the inner shield and confinement building walls. The breaks are needed in order to provide a pathway for the release of radioactivity to the environment. A detailed presentation of our results was recently made at the Heavy Ion Fusion Symposium [7].

The thick-liquid protection provided by the Flibe cross jets and pocket effectively shield the stainless steel structures from damaging and activating neutrons. Using the radioactive afterheat values calculated with ACAB, we performed heat transfer calculations with the CHEMCON code. Despite the loss of all coolant, the temperature of the stainless steel first wall, blanket, and vacuum vessel experience only slight temperature increases. The stainless steel tubes, which serve as the HYLIFE-II first wall, start at 675°C and peak at ~ 679°C only 15 minutes into the excursion. The CHEMCON results were confirmed with those from MELCOR.

There are four main sources of radioactivity that must be considered. First, the x-rays from each target vaporize about 10 kg of Flibe [1]. Although we assume a total LOCA, our analyses conservatively include this Flibe aerosol with its activation products. Second, it is estimated that approximately 140 g of tritium would be trapped within the chamber, blanket, and piping [1]. Tritium migration calculations show that the tritium would be rapidly released from the target chamber during an accident [8]. We assume that entire tritium inventory is converted to the more radiotoxic HTO form. This yields a mass of HTO aerosol of 930 g, which we round up to 1 kg.

Third, we assume that the corrosion of type 304 stainless steel (SS304) by Flibe within the chamber and blanket can be limited to 1 μm/y via corrosion control methods. Additionally, we assume that the Flibe clean-up system can maintain the mobilizable inventory of corrosion products to a 1-y supply. Given a total surface area of 1040 m^2, we obtain a corrosion product inventory of 8.3 kg. Finally, we use data from oxidation-driven mobilization experiments on PCA performed INEEL to calculate an additional 0.5 kg of SS304 (we assume that SS304 mobilization will be the same as that from PCA) that is mobilized under exposure to steam [9,10]. Adding this 0.5 kg to the 8.3 kg of corrosion products, we have ~ 9 kg, which we round up to 10 kg of SS304. Since only ~ 5% of the Flibe is within the chamber and blanket at any time, we transport a mass of 0.5 kg of SS304 aerosol. For the Flibe and SS304 aerosols, we assume a particle size distribution extending from 0.1 to 10 μm in diameter.

In calculating the consequences of the release, we assume typical weather conditions including Class D atmospheric stability, 4 m/s wind speed, 250 m inversion layer, and an initial building wake of 100 m wide × 50 m high. We conservatively assume a ground-level release and take no credit for thermal plume rise. We calculate doses along the plume centerline. Since the release occurs at ground level, the MEI is at the site boundary of 1 km. Our dose goal is 10 mSv (1 rem), which is consistent with the no-evacuation standard set by the Environmental Protection Agency and adopted in the Fusion Safety Standards [11,12].

Using the LOCA scenario, we calculate a total dose of 4.3 mSv (430 mrem) to the MEI. This dose is due almost entirely to the tritium release. Approximately 86% of the HTO is released, while the rest condenses within the building. The Flibe and stainless steel aerosols contribute only 42 μSv (4.2 mrem) and 3.0 μSv (0.3 mrem), respectively. While other accident scenarios must still be considered, it is encouraging that even a full release of tritium would lead to a dose of only ~ 5 mSv (5 rem). Although it goes beyond the intent of the Fusion Safety Standards, the assumption of pessimistic weather conditions (Class F stability and 1 m/s wind speed) yields a higher dose of ~ 56 mSv (5.6 rem).

3.2. Sombrero

when exposed to air or steam. A key issue for the Sombrero safety analysis is the tritium inventory within the chamber and blanket. While the original design study calculated a tritium retention of only 10 g within the first wall and blanket (based on a tritium retention of 5 appm), recent neutron irradiation studies suggest that some damaged carbon composites may retain tritium at much higher levels of 100-1000 appm, which would result in 0.1-1 kg of tritium. Our accident analysis assumes a carbon composite tritium inventory of 1 kg.

Additionally, the xenon atmosphere (at a pressure of ~ 70 Pa) within the target building, which protects the first wall from damaging x-rays and debris, may pose a significant radiological hazard. The dominant isotope, however, is ^{125}I, which may be routinely removed by the chamber vacuum system. Alternatively, a modified version of the Sombrero design may be able to operate with krypton rather than xenon. The activation products from krypton are significantly less hazardous than those produced from xenon.

For a modified Sombrero using krypton for first wall protection, we calculate an MEI dose of 8.3 mSv (830 mrem). The use of xenon would lead to a dose of 9.5 mSv (950 mrem) if the non-xenon activation products can be removed or 54 mSv (5.4 rem) if the iodine and cesium activation products are included in the release. For the krypton case, a total tritium inventory of 1.9 kg can be tolerated before the 10 mSv limit is reached.

4. Driver-Chamber Interface

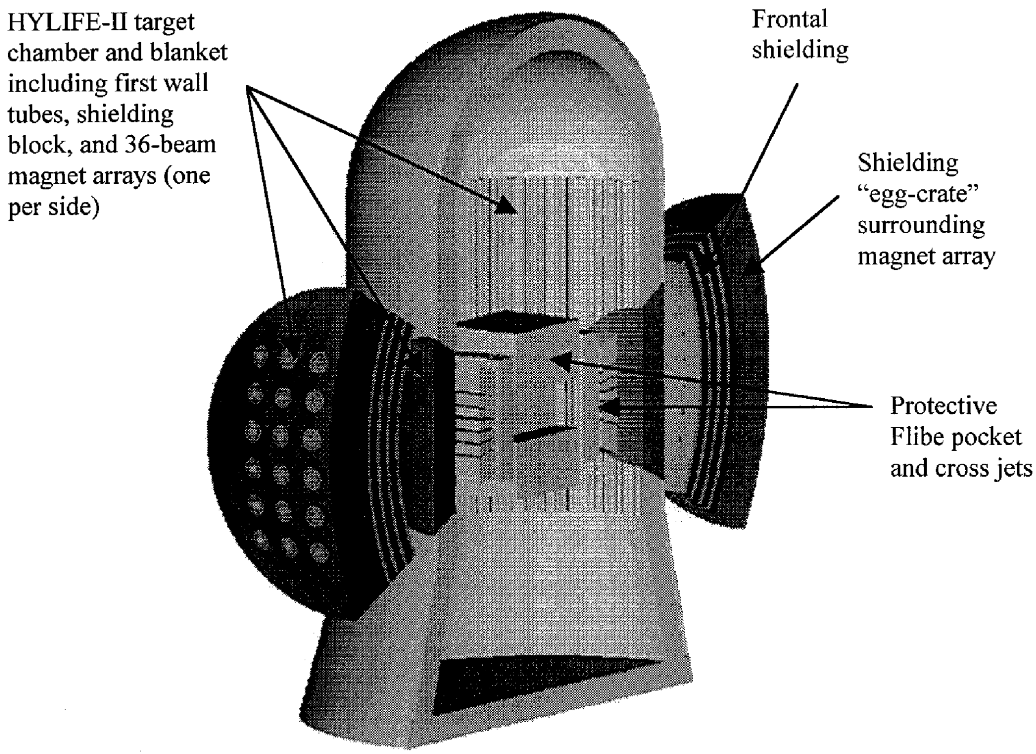
The driver-chamber interface is a concern both for laser- and heavy-ion-driven designs. For lasers, one must contend with radiation damage to the final optical components that sit in the line-of-sight of high-energy neutrons. Additional optical components must be considered as well. For heavy-ions, there are many trade-offs between issues such as focusing length, number of beams, driver cost, chamber transport, and superconducting magnet survivability, maintenance, and activation.

4.1. Heavy Ions

Adequate shielding of final focusing magnets has long been recognized as an important issue in heavy-ion power plant design. Previous designs have been able to effectively shield the magnets rather easily, because the small number of beams (typically 12-20) allowed shielding thicknesses of 30-40 cm on the inner bore of each magnet [14,15]. Cost considerations, however, have driven recent accelerator designs towards a greater number of beams, and thus, less space per beam for magnet shielding [16,17]. This reduction makes the protection of the magnets more of a challenge.

In addition to quench avoidance, key magnet shielding issues are radiation damage, cooling, and neutron activation/waste management. While the magnets will not quench, the radiation-limited lifetimes are unacceptably short [18,19]. In addition, neutron activation of the superconductors is high enough that recycling may be difficult and the waste will not be eligible for disposal via shallow land burial [18,19]. The recirculating power needed for magnet cooling is significant, but appears to be manageable [19]. Recent work (see FIG. 1) mitigates the radiation damage problem through the use of alternate shielding materials, frontal shielding, shielding placed *outside* the magnets, longer focusing lengths, and increased array angles [20]. Magnet lifetimes are now projected to exceed the lifetime of the power plant.

FIG. 1. Recent shielding work has extended the final focusing magnet lifetime to beyond the expected plant lifetime.



4.2. Lasers

For a laser-driven IFE power plant, one must protect the final focusing system, which consists of the final focusing mirrors and final optical element. While these components are adequately protected from x-rays and debris by the gas within the target building, they are susceptible to radiation damage from neutrons and gamma-rays. If the components have short lifetimes, they can become a burden from economic (based upon maintenance and plant availability) and waste management perspectives.

A detailed, 3-D model of the Sombrero laser-driven power plant design was developed to calculate neutron and gamma-ray fluences and doses in the final focusing system [21]. Variations of this model were created for open solid-angle fractions of 0.25% (from the published report) and 5% (increased to provide additional bandwidth for improved beam smoothing). While the original design study utilized grazing incidence metal mirrors (GIMMs) as the final optical element, we replaced them with transmissive fused silica wedges for use with a diode-pumped solid-state laser driver [22].

In our review of the previous work on the Sombrero final focusing system, we found that gamma-ray doses were not considered [2]. Recent work in support of the National Ignition Facility indicates that gamma-ray doses can be of great importance when estimating the lifetime of transmissive optical components [23]. Gamma-ray doses may need to be considered for reflective optics as well. We also found that the final optical elements (GIMMs in the original Sombrero design) were not modeled. Rather, only a 1-D scaling was performed to calculate the fast neutron flux for the final focusing mirrors [2]. Our study shows that this flux was then underestimated and that it is essential to use 3-D modeling in neutronics

Our results indicate that neutrons scattered off of the final optics dominate the fast neutron flux at the final focusing mirror and that this flux is proportional to the open solid-angle fraction, as well as the thickness of the wedges or GIMMs [21]. Lifetime fluence limits have not yet been established, and thus, we estimate the component lifetimes for fluence limits ranging from 10^{20} to 10^{22} n/cm^2. Using these limits, we estimate the fused silica wedge lifetime to be between 0.33 to 33 full-power-years (FPY). In all cases, the wedges would easily qualify for disposal via shallow land burial. For the shortest lifetimes, however, the waste volumes could become significant. It is worth noting that the wedges may be operated at elevated temperature and that it is possible that they will be self-annealing [21].

The final focusing mirrors would have dielectric coatings and are thought to be more sensitive to radiation damage [2]. Assuming a fast neutron fluence limit of 10^{18} to 10^{19} n/cm^2 and estimate the lifetime of these elements would range from 0.25 to 2.5 FPY. Experimental data on radiation damage, survivability, annealing, and recycling of optical components are needed. Such data should include not only neutrons but gamma-rays as well.

5. Target Materials

The selection of IFE target materials needs to be made with S&E considerations. The ability to recycle target materials, waste management after use, and accident safety all must figure in the choice of material [24]. Due to the small quantity of material involved (< 0.1 m^3), we assumed that isotopic separation may be an economically viable option and completed a survey of 264 stable isotopes from lithium to polonium [24]. Our criteria are the following: a contact dose rate limit of 114 Gy/h at a time of 7 days after irradiation (yields an integrated dose of 30 MGy over 30 years), a waste disposal ratings (WDR) of less than unity (should enable ultimate disposal via shallow land burial), and an accident dose of less than 10 mSv (1 rem). Since detailed accident analyses are not yet available, a simplified set of release fractions were utilized [24].

Out of 264 stable isotopes, 138 were able to meet the contact dose rate criterion, while 176 had a WDR less than unity. Ninety-seven isotopes were able to meet both the contact dose rate and WDR criteria simultaneously. Of these 97 isotopes, 48 are also able to meet the accident dose criterion [24].

Of special note is the fact that mercury, lead, hafnium, ytterbium, neodymium, and palladium would require relatively minor isotopic separation to meet all three criteria (36-99% of the isotopes that make up the natural element meet all three criteria). Only ^{204}Pb, for example, is problematic for natural lead. Once realistic release fractions are considered, it is expected that even more isotopes and elements will meet the S&E criteria.

6. Fast Ignition

Power plant designs that make use of fast ignition offer many exciting opportunities. In addition to reducing the required driver energy and cost of electricity, fast ignition appears to offer relaxed target fabrication requirements, the possibility of using advanced fuels with significant S&E and target fabrication characteristics, or higher gains with lower repetition rates [25,26].

Tritium-lean targets are an example of advanced fuels that have exceptional S&E characteristics [27-29]. Using fast ignition, such targets may be viable at reasonable driver small "seed" of equimolar D-T fuel is used, the overall tritium fraction falls to < 1%, and the targets may be entirely self-sufficient from a tritium breeding point-of-view [28]. This greatly opens the available parameter space, and frees power plant designers from the need to breed tritium in the blanket. Designers can select coolants to satisfy other criteria such as pumping power, beam propagation, and power conversion [29]. Using an alternate liquid such as B_2O_3, thick-liquid protection schemes work with thinner blankets and radiation damage rates and neutron activation are reduced [29].

7. Collaborations

An important part of our work is accomplished via domestic and foreign collaborations. A good example of this is the collaboration with our colleagues in the Fusion Safety Program at INEEL. INEEL personnel have extensive experience in fusion safety. They have developed and benchmarked computer codes, designed and built experiments, and performed safety analysis for magnetic fusion experiments such as the International Tokamak Experimental Reactor and many power plant designs. We have adopted and modified the CHEMCON code from INEEL, and we use the INEEL-modified (for fusion) version of MELCOR. LLNL personnel contribute to discussions regarding new experiments such as the Fusion Liquid Release Experiment, FLIQURE. We regularly consult with INEEL personnel regarding assumptions and methods for safety analysis.

Our neutron activation code, ACAB, was developed and is continually being improved by Professor Javier Sanz at the University of Madrid's Department of Industrial Engineering (and the Institute of Nuclear Fusion). Professor Sanz and his team contribute to our efforts with computer codes and data libraries. Our recent safety assessments, for example, make use of an accident dose library, which the Spanish team created with the MACCS2 accident consequences code [7,24].

8. Future Work

Our ongoing and future work will continue to be dominated by safety analysis. We are analyzing additional accidents for the HYLIFE-II and Sombrero power plant designs. An analysis for a generic target fabrication facility is underway and will include updated estimates of the tritium inventory and transport of activated target materials.

We will begin working with the ARIES Team during the latter part of fiscal year 2000. The ARIES Team performs integrated assessments of fusion concepts, which include relevant physics, neutronics, component design, systems engineering, economics, and safety analyses. The ARIES Team will study IFE systems starting with dry wall concepts, such as Sombrero. LLNL participation will be included in the areas of neutronics, safety, and systems engineering.

Work on fast ignition concepts should begin in fiscal year 2001. This work will leverage off of recent Defense Programs-funded research into fast ignition target design. We hope to quantify the power plant implications of fast ignition.

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University of California
Lawrence Livermore National Laboratory
Technical Information Department
Livermore, CA 94551